NON-PUBLIC?: N
ACCESSION #: 8910120144
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Nine Mile Point Unit 2 PAGE: 1 OF 4

DOCKET NUMBER: 05000410

TITLE: Manual Reactor Scram Due to Equipment Failure And Entry Into Restricted Zone
EVENT DATE: 09/08/89 LER #: 89-024-00 REPORT DATE: 10/05/89

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 088

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Gary Thompson, Supervisor TELEPHONE: (315) 349-2708
System Support & Testing

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: SJ COMPONENT: JX MANUFACTURER: L045
REPORTABLE NPRDS: Yes

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

At 18:00:45 hours on September 8, 1989, with the Reactor Mode Switch in "RUN of and the Reactor at 88% rated thermal power (930 MWe), Nine Mile Point #2 experienced a downshift of Reactor Recirculation Pumps to slow speed. This placed the unit above the 100% rod line with core flow <45% (restricted area of operation). The control room annunciators notified the operations personnel of the downshift. Operations personnel immediately performed the required actions per N2-OP-29, Reactor Recirculation System, by placing the reactor mode switch to "SHUTDOWN", initiating a reactor scram. Operations personnel proceeded with scram recovery per N2-OP-101C, Plant Shutdown. 10CFR50.72 notification was made on September 8, 1989, at 2013.

The root cause for recirculation pump trip is a failure of a 24 volt DC power supply (C33A-K613) within the Feedwater Control System. Corrective

action was to replace the faulty power supply.

END OF ABSTRACT

TEXT PAGE 2 OF 4

I. DESCRIPTION OF THE EVENT

On September 8, 1989, at 18:00:45, Nine Mile Point #2 experienced an inadvertent downshift of the Reactor Recirculation Pumps to slow speed. At the time of the event the Reactor was at 88% thermal power (930 MWe), 83% of rated core flow and above the 100% rod line. Due to the downshift, core thermal power and core flow decreased, stabilizing approximately 30 seconds later at 447. rated thermal power and 34% of rated core flow. The computer indicated a "Recirc A Low Level Trip" and "Recirc B Low Level Trip" (indicative of a Low Low, 108.8 inches, Reactor Level), "Low Power Alarm Point" and "Low Power Setpoint" (indicative of a Low Steam Flow Signal to the Rod Worth minimizer), and it is believed that the "C" feedwater control valve lockout relay energized at this time and latched. Valid rod blocks were initiated per design. No actual level or steamflow transients were experienced at this point in the event.

Two seconds later, the Feedwater System Controller Signal Failure and Reactor High Pressure alarms came in with the Reactor water level analog computer point reading 191 inches. The Reactor water level was high due to the swell caused by the rapid power reduction.

Upon being alerted to the recirculation pump downshift by the control room annunciators, the operations personnel took action as directed by N2-OP-29, Figure 3, Section H.8.0 and at 18:01:57, the Reactor Mode Switch was placed to the "SHUTDOWN" position. When the mode switch passed through the "STARUP" position, an AUTO Reactor Protection System initiation occured due to Average Power Range Monitors (APRMs) being greater than 15% in the "STARTUP" mode. A licensed control room operator monitored Feedwater Flow Control to verify proper Reactor water level.

During the event there were no unexpected plant system responses. There were no indications of power oscillations on APRM's while operating inside the restricted zone on the power flow map. The turbine bypass valves did not open during the event due to the power reduction caused by the downshift of the Reactor Recirculation Pumps. The turbine tripped approximately 1:25 minutes after the reactor scram due to the anti-motoring trip. Reactor water level momentarily dropped below the Level 3 setpoint (159.3 inches) and was recovered to approximately 190 inches, and then was controlled at 165 inches. Per design, there were no

Emergency Core Cooling System (ECCS) actuations.

II. ANALYSTS OF THE EVENT

This event is reportable in accordance with 10 CFR 50.73 (a)(2)(iv). "Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)". The downshift to slow speed by recirculation pumps P1A and P1B placed the unit above the 100% rod line with core flow less than 45% (restricted area of operation), requiring a manual Reactor scram by placing the mode switch to "SHUTDOWN" in accordance with Operating Procedure N2-OP-29.

There were no ECCS actuations during the transient, this transient was significantly less severe than the bounding accident analyzed in the Updated Safety Analysis Report chapter 15 "Decrease in Reactor System Flow Rate".

A manual reactor scram is a conservative event and poses no adverse safety consequences at any reactor power. This event did not adversely affect any safety system nor the operators ability to achieve safe shutdown.

III. CAUSE OF THE EVENT

A root cause analysis was performed using Site Supervisor Procedure S-SUP-1, "Root Cause Evaluation Program". The root cause for this event was determined to be failure of the Feedwater Control System logic power supply, C33A-K613. (Equipment failure)

IV. CORRECTIVE ACTION

After review of the plant digital computer printouts and analog computer printouts it was determined that there were a number of digital points that were invalid. The invalid alarms from the event and the recirculation pump downshift were found to all be associated with feedwater circuit 2FWSN34. Two work requests were initiated in order to troubleshoot feedwater circuit 2FWSN34.

The feedwater Level 3 transmitter circuits were tested and found to be operating properly. The "A" transmitter was found to be set slightly high but was adjusted without proble
. It was not high enough to have
any effect on this event.

The feedwater control signal lockout relay circuits were tested and found to be operating as designed. (The "C" feedwater lockout relay had operated during the event).

Each feedwater control signal monitoring relay and circuit was tested and found to be operating properly. (The relays trip their associated lockout relay). A Feedwater System Controller Signal Failure alarm had come in during the event. This .alarm is initiated from the feedwater control signal monitoring relays.

The three 24 volt DC power supplies that feed power to the (2FWSN34) feedwater circuit were checked for any possible problem. Two of the three power supplies were found to have pure DC waveshapes with no ripple. One power supply (C33A-K613) had approximately 4-5% ripple on the output. In the course of troubleshooting this power supply it totally failed - its output went to zero. The power supply was replaced with a new one and it was checked for ripple. The replaced DC power supply had a pure DC waveshape.

This is the only failure of a Lambda power supply to date, so no other power supplies were checked. The failed power supply was sent out for failure analysis to determine if any further corrective action is required.

The loads fed by the failed power supply were reviewed for impact if the power supply had failed and recovered in service. Every invalid alarm and each of the unexplained relay initiations from the event were fed from the failed power supply.

Corrective action required has been accomplished by replacing the failed power supply C33A-K613.

V. ADDITIONAL INFORMATION

A. Previous Similar Events (none)
B. Component Identification

COMPONENT SYSTEM IEEE-805 SYS IEEE-803 COMPONENT

Turbine Bypass Valves TG ISV
Feedwater Control Valve SJ SCV
Lockout Relay N/A 86
Reactor Recirculation Pump AD P

Reactor Protection System JC N/A

ATTACHMENT 1 TO 8910120144 PAGE 1 OF 1

NIAGARA NMP 56845
MOHAWK

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October 6, 1989

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

RE: Docket No. 50-410
LER 89-24

Gentlemen:

In accordance with 10CFR50.73, we hereby submit the following Licensee Event Report:

LER 89-24 Is being submitted in accordance with 10CFR50.73(a)(2)(iv), "Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)".

A 10CFR50.72(b)(2)(ii) report was made at 2013 hours on September 8, 1989.

This report was completed in the format designated in NUREG-1022, Supplement 2, dated September 1985.

Very truly yours,

J. L. Willis
General Superintendent
Nuclear Generation

JLW/DS/lmc
(0961V)

Attachments

xc: Regional Administrator, Region I
Sr. Resident Inspector, W. A. Cook

*** END OF DOCUMENT ***